Transactions of the Korean Nuclear Society Autumn Meeting
Goyang, Korea, October 24-25, 2019

Preliminary Thermal Design Analysis of the Planned 250 KW TRIGA Philippine Research Reactor 1

Julius Federico M. Jecong [a,b*], Sweng Woong Woo [b],
[a] *Philippine Nuclear Research Institute, DOST, Commonwealth Ave., Diliman, Quezon City, 1101, Philippines*
[b] *Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong-gu, Daejeon 34142, Republic of Korea*
[*] *Corresponding author: jmjecong@kaist.ac.kr*

1. Introduction

The Philippine Nuclear Research Institute (PNRI) has operated the only research reactor in the country, Philippine Research Reactor 1 (PRR-1) that was acquired about 65 years ago (Figure 1).

The reactor is an open-pool general-purpose type designed with plate type aluminum MTR fuel. The entire instrumentation system was replaced with modern solid-state components in 1982. During that time, the rest of the reactor were aging and becoming difficult to maintain. As a partial solution, PRR-1 was converted to a 3 MW TRIGA-type reactor in 1988. Unfortunately, just a few weeks after the PRR-1 was successfully tested as a TRIGA reactor, the reactor pool liner sprung a serious leak. Resumption of regular operation had to be suspended pending the repair of the leak. PRR-1 has not been in operation since then.

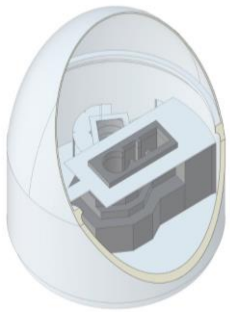

Fig. 1. Philippine Research Reactor (PRR-1)

An on-going project in PNRI intends to reuse the TRIGA fuel in a subcritical reactor assembly [1]. Looking ahead, the effort of PNRI in strengthening and advancing nuclear science and engineering expertise in the Philippines continues with the plan to establish 250 KW TRIGA Research Reactor (RR) in the near future.

This study aims to investigate the thermal hydraulic behavior of the planned 250 KW TRIGA RR using MARS-KS calculation. The design should be capable of safely removing the heat generated in the fuel without producing excessive fuel temperatures or steam void formations, and without closely approaching the hydrodynamic critical heat flux (CHF), under either steady-state operating conditions, anticipated operational occurrences or accident conditions. CHF is the phenomenon in which the efficiency of heat transfer from fuel to coolant is greatly reduced that may lead to abrupt increased in fuel temperature, and potential fuel failure. Unfortunately, since TRIGA RR operate at low pressure, low flow, and natural-circulation conditions, no CHF correlation that existed are directly applicable. The correlation done by Bernath [2] was used to predict CHF in the study. Although considered outside its applicable range, this correlation was used in safety analysis report of 3 MW TRIGA PRR-1 [3] and safety analysis of University of Wisconsin TRIGA reactor [2]. The preliminary analysis on this paper is limited with the steady-state thermal performance of hottest channel.

2. Materials and Methods

In this section, the planned reactor pool tank dimensions, and TRIGA fuel geometry are briefly introduced. Then, the MARS input model for the design is presented, which is refined by reflecting the results of preliminary calculations.

2.1 PRR-1 TRIGA fuel

The PRR-1 TRIGA fuel is a low-enriched, long-lifetime homogeneous alloy of uranium and zirconium-hydride ($UZrH_{1.6}$) fuel. This alloy provides both fuel and moderator necessary to achieve and sustain criticality. The intimate mixing of hydrogen and fuel material provides the TRIGA core a large prompt negative temperature coefficient of reactivity.

The active fuel section is about 50.8 cm long, has a 2.97 cm diameter, and are contained in a 0.05 cm thick Incoloy-800 cladding. A zirconium rod with a diameter of 0.63 cm is located through the center of the active fuel section. The fuel rod is 75.18 cm long and 3.07 cm in diameter as shown in Figure 2.

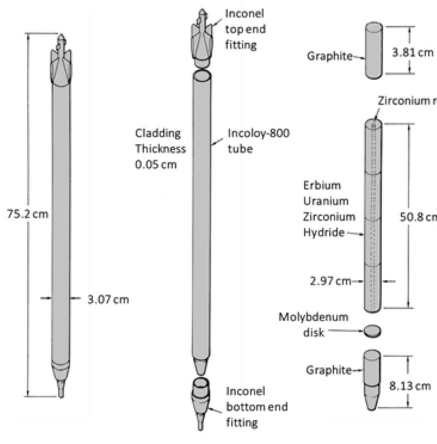

Fig. 2. PRR-1 TRIGA fuel geometry

2.2 Reactor pool and reactor core

The planned reactor pool is cylindrical water tank with diameter of 2.8 m and a depth of 4.51 m. The reactor core which is elevated 0.60 m from the bottom of the reactor pool is 0.67 m high as shown in Figure 3.

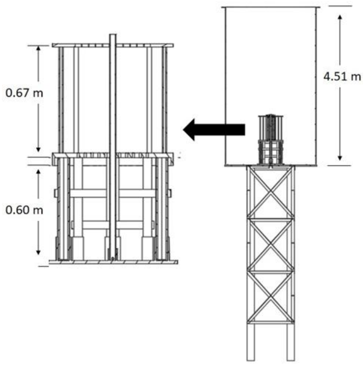

Fig. 3. Planned reactor pool and reactor core.

2.3 MARS-KS Simulation

The MARS-KS (Multi-dimensional Analysis of Reactor Safety KINS Standard) code was used to preliminary estimate and evaluate the thermal hydraulic behavior of the planned research reactor in steady-state.

Previous calculation suggests that around 66 fuels will be used to achieve 250 KW research reactor, which means 3787.87 W per element on the average. The boundary conditions were modeled using the time dependent volumes. The reactor core flow inlet is modeled by time-dependent junction component connected to time-dependent volume component. On the other hand, the reactor core flow outlet is modeled by single-junction component connected to time-dependent volume. Other thermal hydraulic parameters used in calculation are shown in Table 1.

Table I: Conditions in simulation

Parameters	Value
Pressure	
lower plenum	139.55 kPa
core fuel (average)	137.50 kPa
upper plenum	133.00 kPa
Core Inlet Temperature	27 ºC
Average Power	3787.87 W
Maximum Radial Peaking Factor	2.0
Maximum Axial Peaking Factor	1.3
Fluid Velocity	0.12 m/s

Reactor core should be designed with appropriate margin to assure that acceptable fuel design limits are not exceeded during any condition. The maximum heat flux anywhere in the core is limited by the DNB ratio in a water-cooled reactor, therefore conservative values of radial peaking, and a chopped cosine shape axial power profile were chosen to have a high safety margin. Figure 4 shows the axial power distribution used in the fuel.

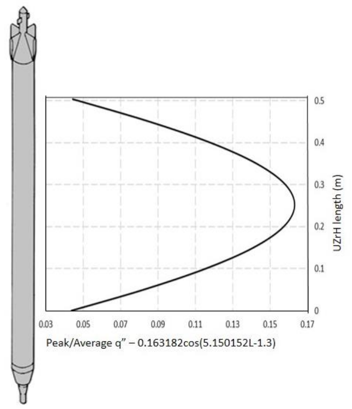

Fig. 4. Axial power profile of TRIGA Fuel rod

The fuel thermal conductivity k [4] and specific heat capacity C_p [5] as a function of temperature equations of $UZrH_{1.6}$ fuel that were used in the code are given by

$$k(T) = 1.11x10^{-3}T + 21.23 \text{ [W/m-K]} \tag{1}$$

$$C_\rho = 363 + 0.75T \text{ [J/kg-K].} \tag{2}$$

The fuel density (ρ = 8.41 x 10^3 [kg/m^3]) which was used for volumetric heat capacity calculation (ρCp) was derived by Negut et.al. [6]. To ensure the correct simulation of the flow and heat transfer, the input code was checked against experimental data of Avery et.al [2].

2.4 Bernath CHF Correlation

To predict CHF, Bernath CHF correlation is used, expressed as equations below

$$q_c = h_c\left(T_{w_c} - T_B\right), \tag{3}$$

$$h_c = \left[10890\left\{\frac{D_e}{D_e + D_i}\right\} + s\upsilon\right]5.678263, \tag{4}$$

$$s = \begin{cases} \dfrac{48}{D_e^{0.6}} & D_e \le 0.1ft, \\ 90 + \dfrac{10}{D_e} & D_e > 0.1ft, \end{cases} \tag{5}$$

$$T_{W_c} = 57\ln P - 54\left\{\frac{P}{P+15}\right\} - \frac{\upsilon}{4}, \tag{6}$$

where q_c is the critical heat flux [W/m^2], h_c is critical coefficient of heat transfer [W/m^2-ºC], T_{wc} is critical fuel surface temperature [ºC], T_B is bulk fluid temperature at the location of CHF [°C], D_e is the hydraulic diameter [ft], D_i is the heated diameter (heated perimeter divided by π) [ft], P is the pressure [Psia], and υ is the fluid velocity [ft/s]. Based on this equations, in subcooled boiling, the CHF is a function of the fluid velocity, the degree of subcooling, and the pressure. This correlation was developed from a circular, rectangular, and annular channels, pressure of 0.1 to 20.7 MPa, mass flux range from 750 to 15000 kg/m^2-s and hydraulic diameter of 0.36 to 1.7 cm.

3. Results and Discussion

This section presents the preliminary results of the thermal design analysis of the planned 250 KW TRIGA PRR-1. Figure 5 shows the temperature distribution of the fuel centerline, cladding, and coolant along the axial distance of hottest channel. As expected, the cladding temperature is lower than the fuel centerline temperature. Even with conservative radial and axial peaking factors, the peak of fuel centerline temperature is around 347 K. This means that the calculated peak temperature is far below the fuel excessive-swelling temperature limit of 1023 K [3]. The bulk coolant temperature increases steadily from its entering value of is 301 K at the bottom of the core then decreases before it reach 320 K as it exits the core.

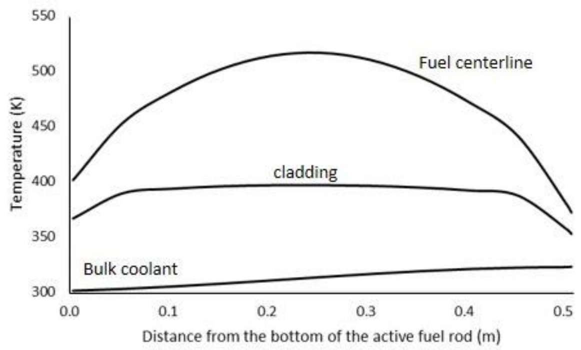

Fig. 5. Axial temperature for the hot channel

In Figure 6, the lower curve shows the actual heat flux into the channel. The peak heat flux occurs at the axial center of the fuel rod, and the calculated DNB ratio (CHF divided by actual flux q’’) is minimum at this location. No nucleate boiling is predicted for the hottest rod over the length of 0.508 m. Since it is a very low power research reactor, the minimum DNB ratio of 6.18 was calculated. Determining the value of CHF is a must because this knowledge will be used to prevent the reactor from operating near DNB point. Although it was demonstrated recently [2] that similar CHF behavior is to be expected from natural and forced circulation at a low-flow condition, it is still important to determine the values of CHF at different flow velocity.

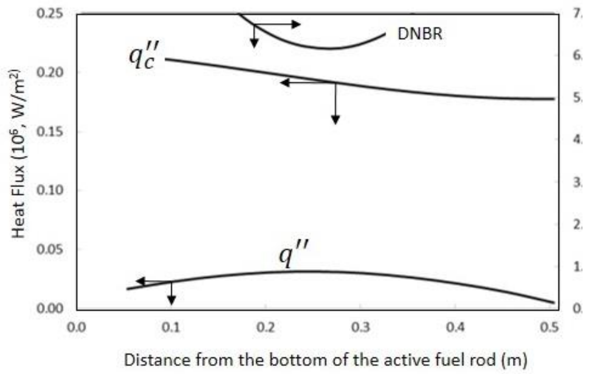

Fig. 6. Thermal analysis of the hottest channel.

Good agreement between the simulation results and experimental data of Avery et.al [7] ensured the validity of the model.

4. Conclusions

Steady-state thermal design analysis of the planned 250 KW TRIGA PRR-1 was performed using MARS-KS code. Preliminary results suggest that the reactor is relatively safe because it is far from the occurrence of the DNB, and reaching CHF. For the next phase of work, instead of estimation, the actual power profile of the hottest fuel will be used for thermal analysis. More analysis will be done such as effect of flow rate, effect of the hot rod factor, effect of the inlet temperature, and transient analysis to understand more the thermal characteristics of the research reactor design.

REFERENCES

[1] A. Asuncion-Astronomo, Ž. Štancar, T. Goričanec, and L. Snoj, “Computational design and characterization of a subcritical reactor assembly with TRIGA fuel,” Nucl. Eng. Technol., vol. 51, no. 2, pp. 337–344, 2019.
[2] M. Avery, J. Yang, and M. Corradin, “Critical Heat Flux in TRIGA-Fueled Reactors Cooled by Natural Convection,” Nucl. Sci. Eng., vol. 172, no. 3, pp. 249–258, 2014.
[3] Philippine Atomic Energy Commission, “Safety Analysis Report for the 3-MW Forced Flow Operation TRIGA Conversion Philippine Research Reactor,” 1987.
[4] A. Z. Mesquita and H. C. Rezende, “Experimental Thermal-Hydraulic Analysis of the IPR-R1 TRIGA Nuclear Reactor,” in Proceedings of ENCIT 2006 -- ABCM, Curitiba, Brazil, 2006, no. 1.
[5] D. Stahl, “Fuels for Research and Test Reactors, Status Review: July 1982,” Argonne National Laboratory, 1982.
[6] G. Negut, M. Mladin, I. Prisecaru, and N. Danila, “Fuel behavior comparison for a research reactor,” J. Nucl. Mater., vol. 352, no. 1–3, pp. 157–164, 2006.
[7] J. Yang et al., “Critical heat flux at conditions representative of TRIGA-type reactors – single, three rod and four rod bundle CHF data,” Department of Energy National Nuclear Security Administrtion, 2012.